# 12. RADIATION PROTECTION

## 12.1 Introduction

The economic simplified boiling-water reactor (ESBWR) design control document (DCD) Tier 2, Chapter 12, “Radiation Protection,” describes the kinds and quantities of radioactive materials expected to be produced in the operation of the ESBWR reactor and the means for controlling and limiting radiation exposures within the requirements in Title 10, Part 20, “Standards for Protection Against Radiation,” of the *Code of Federal Regulations* (10 CFR Part 20). The ESBWR reactor design incorporates radiation protection measures intended to ensure that internal and external radiation exposures to station personnel, contractors, and the general population, resulting from plant conditions, including anticipated operational occurrences (AOOs), will be within regulatory criteria and will be as low as reasonably achievable (ALARA).

The U.S. Nuclear Regulatory Commission (the NRC or staff) evaluated the information in Chapter 12 of the ESBWR DCD against the criteria in Chapter 12 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition” (SRP). Compliance with these criteria provides assurance that doses to workers will be maintained within the occupational dose limits of 10 CFR Part 20. These occupational dose limits, applicable to workers at NRC-licensed facilities, restrict the sum of the external whole-body dose (deep-dose equivalent) and the committed effective equivalent doses resulting from radioactive material deposited inside the body (deposited through injection, absorption, ingestion, or inhalation) to 50 millisievert (mSv) (5 rem) per year with a provision (*i.e.*, by planned special exposure) to extend this dose to 100 mSv (10 rem) per year with a lifetime dose limit of 250 mSv (25 rem) resulting from planned special exposures.

The SRP acceptance criteria also provide assurance that radiation doses resulting from exposure to radioactive sources both outside and inside the body can be maintained well within the limits of 10 CFR Part 20 and ALARA. The balancing of internal and external exposure necessary to ensure that the sum of the doses is ALARA is an operational concern. An applicant seeking a combined license (COL) must address these operational concerns, as well as programmatic radiation protection concerns.

## 12.2 Ensuring That Occupational Radiation Doses Are ALARA

### 12.2.1 Regulatory Criteria

The applicable criteria and guidance include the following:

- 10 CFR 20.1101, “Radiation Protection Programs,” and 10 CFR 20.1704, “Further Restrictions on the Use of Respiratory Protection Equipment”

- 10 CFR 50.34(b)(3), as it relates to the kinds and quantities of radioactive materials produced and the means for controlling and limiting radiation exposures within the limits of 10 CFR Part 20.

- Regulatory Guide (RG) 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” Revision 3, May 2000

- RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” Revision 3, November 1978

- RG 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable,” Revision 3, June 1978

- RG 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable,” Revision 1-R, May 1977

### 12.2.2 Summary of Technical Information

In addition to providing radiation exposure limits for workers and members of the public, 10 CFR 20.1101(b) requires that, to the extent practical, procedures and engineering controls based on sound radiation protection principles be employed to achieve occupational doses and doses to the public that are ALARA. In addition, 10 CFR 20.1704(a) requires that the intake of airborne radioactive materials be consistent with maintaining total effective dose equivalent ALARA. RG 8.8 provides specific guidance and criteria on the design, construction, and operation of a nuclear power plant to meet this regulatory requirement. Programmatic and policy considerations associated with plant operations that are needed to assure that radiation doses will be ALARA (as discussed in RGs 8.8, 8.10, and 1.8) are outside the scope of this design certification. The applicant has identified COL action items (see Section 12.2.3.1 below) to ensure that license applicants referencing the ESBWR design will address these issues.

### 12.2.3 Staff Evaluation

The staff reviewed the information in DCD Tier 2, Section 12.1, “Ensuring That Occupational Radiation Exposures Are ALARA,” to assess adherence to the guidelines in RG 1.70, as well as the criteria in Section 12.1 of the SRP regarding the radiation protection aspects of the ESBWR reactor design. Specifically, the staff reviewed Section 12.1 of DCD Tier 2 to ensure that the applicant had either committed to adhere to the criteria of the regulatory guides and staff positions referenced in Section 12.1 of the SRP or had provided acceptable alternatives.

#### 12.2.3.1 Policy Considerations

In DCD Tier 2, Section 12.1.1, “Policy Considerations,” the applicant described the design, construction, and operational policies that ensure that ALARA considerations are factored into each stage of the ESBWR design process. The applicant has committed to ensure that the ESBWR plant will be designed and constructed in a manner consistent with the guidelines of RG 8.8. In particular, DCD Tier 2, Section 12.1.1.1, “Design and Construction Policies,” states that the ALARA philosophy was applied during the initial design of the ESBWR. The plant design was reviewed in detail for ALARA considerations and modified as necessary during the design phase. Experience related to ALARA performance gained from operating plants was continuously integrated during the design phase of the ESBWR standard plant. This ALARA policy is consistent with the guidelines of RG 8.8 and is therefore acceptable.

The requirements of 10 CFR Part 20 specify that all licensees must develop, document, and implement a radiation protection program. Specifically, this program shall encompass the

ALARA concept and provide for maintaining radiation doses and intakes of radioactive materials ALARA. The operational ALARA policy forms the basis for the operating station's ALARA manual. The detailed policy considerations regarding overall plant operations and implementation of such a radiation protection program are outside the scope of this design certification review.

To maintain doses to plant personnel ALARA, the applicant stated, in DCD Tier 2, Section 12.1.4, "COL Information," that the COL applicant will present, consistent with the criteria in RG 1.70, the operating procedures and techniques it will implement to ensure that occupational radiation doses are ALARA (**COL Action Item 12.1.4.3**). In addition, a COL applicant referencing the ESBWR certified design will demonstrate how its operational ALARA policy conforms to the requirements of 10 CFR Part 20 and the recommendations of Revision 2 to RG 1.8 (**COL Action Item 12.1.4.2**), RG 8.8 (**COL Action Item 12.1.4.4**), and Revision 1-R to RG 8.10 (**COL Action Item 12.1.4.1**).

12.2.3.2 Design Considerations

The plant radiation protection design should ensure that individual doses and collective total effective dose equivalent (person-rem) to plant workers and to members of the public are ALARA and that individual doses are maintained within the limits of 10 CFR Part 20. DCD Tier 2, Section 12.1.2, "Design Considerations," describes the objectives for the general design and shielding. Specifically, Section 12.1.2 states that the basic management philosophy guiding the ESBWR design is to ensure that exposures are ALARA by designing structures, systems, and components to achieve the following objectives:

- minimize the necessity for and the amount of time spent in radiation areas

- minimize radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention

The staff finds that these design objectives are consistent with the guidelines in RG 8.8.

Section 12.1.2 of DCD Tier 2 describes several design features that satisfy the objectives of the plant's radiation protection program. Examples of these features include the following:

- To the extent practicable, materials in contact with the reactor coolant system (RCS) have low concentrations of cobalt and nickel. This reduces the amounts of Co-60 and Co-58 introduced in the RCS. (Co-60 and Co-58 are the major sources of radiation exposure during shutdown, maintenance, and inspection activities at light-water reactors (LWRs).)

- Central control panels (*i.e.*, the control rod drive (CRD) maintenance control panel and the reactor building sample panel) are in separate, shielded rooms with low-radiation background.

- Adequate spacing and laydown areas facilitate access for maintenance and inspection. Separate low background rooms are provided for CRD and hydraulic control unit maintenance.

- The time spent in radiation areas will be minimized through enhanced servicing convenience for anticipated maintenance or potential repairs, including ease of disassembly of components for replacement or removal to a lower radiation area for repair or servicing.

- Radioactive systems are separated from nonradioactive systems, and high-radiation sources are located in separate shielded cubicles.

- Equipment requiring periodic service or maintenance (*e.g.*, pumps, valves, and control panels) is separated from more radioactive sources (*i.e.*, tanks and piping).

- Valves located in high-radiation areas are equipped with reach rods or motor operators to minimize operator exposure.

- Equipment and piping are designed to minimize the accumulation of radioactive materials.

- Drains are located at low points of systems and components.

- Piping is seamless, and the number of fittings is minimized, thereby reducing the radiation accumulation at seams and welds.

- Use of flushing connections minimizes the buildup of crud in system components.

- Adequate space and means are provided for the use of movable radiation shielding to provide personnel protection from radioactive sources, when required.

These design considerations incorporate the basic management philosophy guiding the ESBWR design effort and are consistent with the guidelines in RG 8.8.

In addition to the features described above, the ESBWR reactor design incorporates several features that represent improvements over many currently operating plants:

- The ESBWR design uses natural circulation, resulting from thermal convective forces in the reactor vessel, to circulate coolant through the core. This design eliminates the need for reactor water recirculation system piping and associated active pumps and valves which historically have been significant sources of personnel exposure in current boiling-water reactor (BWR) designs.

- Material selection for the ESBWR design includes minimizing the use of cobalt-bearing components in the reactor water systems. In addition, the ESBWR main condenser has titanium or stainless steel tubes and tubesheets to minimize service water in-leakage and the resultant activation of reactor water contaminants.

- The condensate system in the ESBWR uses hollow-fiber-filled filters that require approximately half the maintenance of typical systems.

- The low-pressure feedwater drains from the feedwater heaters are cascaded back to the condenser; thus, all corrosion products from these drains are filtered via condensate filters/demineralizers before returning to the reactor pressure vessel to minimize the activation of these materials.

In request for additional information (RAI) 12.2-19, the staff asked the applicant to verify that the shielding around the reactor vessel is sufficient to allow personnel access to the upper drywell during fuel-handling operations. After reviewing the applicant's response to this RAI, the staff issued the following Supplement No. 1 to RAI 12.2-19 concerning the burnup value of the fuel assembly used in the shielding analysis in the applicant's response:

> The shielding analysis provided in RAI 12.2-19 is based on 35 GWd/MTU exposure of a GE-14 fuel bundle. However, Figure 4A-18e lists several bundles in the core with burn-ups greater than 35 GWd/MTU (*e.g.,* the bundle at 5,17 has a maximum average exposure of 50.38). In addition, Topical Report NEDC-33242P (currently under review in concert with the ESBWR design review) indicates a peak bundle average exposure of 67 GWd/MTU. Clarify how the RAI response provides a bounding analysis of the operational event in question.

Subject to resolution of RAI 12.2-19, Supplement No. 1, this remains **Open Item 12.2-19**.

The design features described in Section 12.1.2 of DCD Tier 2 are intended to minimize personnel exposures and comply with the guidelines of RG 8.8. As such, these design features should maintain individual doses and total person-rem doses to plant workers and to members of the public ALARA, while maintaining individual doses within the limits of 10 CFR Part 20.

### 12.2.3.3 Operational Considerations

Operational considerations regarding the implementation of a radiation protection program are outside the scope of this design certification review. Chapter 12 of the SRP lists the following regulatory guides that pertain to DCD Tier 2, Chapter 12:

- RG 8.2, "Guide for Administrative Practices in Radiation Monitoring," February 1973

- RG 8.7, "Instructions for Record Keeping and Recording Occupational Radiation Exposure Data," Revision 2, November 2005

- RG 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," Revision 1, July 1993

- RG 8.13, "Instruction Concerning Prenatal Radiation Exposure," Revision 3, June 1999

- RG 8.15, "Acceptable Programs for Respiratory Protection," Revision 1, October 1999

- RG 8.20, "Applications of Bioassay for I-125 and I-131," Revision 1, September 1979

- RG 8.25, "Air Sampling in the Work Place," Revision 1, June 1992

- RG 8.26, "Applications of Bioassay for Fission and Activation Products," September 1980

- RG 8.27, "Radiation Protection Training for Personnel at Light-Water Cooled Nuclear Power Plants," March 1981

- RG 8.28, "Audible-Alarm Dosimeters," August 1981

- RG 8.29, "Instructions Concerning Risks from Occupational Radiation Exposure," Revision 1, February 1996

- RG 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," July 1992

- RG 8.35, "Planned Special Exposures," June 1992

- RG 8.36, "Radiation Dose to the Embryo/Fetus," July 1992

- RG 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," Revision 1, May 2006

Addressing the above regulatory guides is outside the scope of this design certification review. In DCD, Tier 2, Section 12.1.3, the applicant stated that the COL applicant will address operational considerations of the SRP consistent with the level of detail provided in RG 1.70, including a description of how the COL applicant will comply with the recommendations of (or provide acceptable alternatives to) the preceding regulatory guides (**COL Action Item 12.1.4.3**).

### 12.2.4 Conclusions

Due to the open item that remains to be resolved for this section, the staff was unable to finalize its conclusion regarding acceptability. These design features are intended to maintain individual doses and total person-rem doses to plant workers and to members of the public ALARA, while maintaining individual doses within the limits of 10 CFR Part 20.

As previously stated, the COL applicant will address the policy and operational considerations for the ESBWR. The staff finds it acceptable to defer the discussion of the material addressed by COL Action Items 12.1.4.1, 12.1.4.2, 12.1.4.3, and 12.1.4.4 until this time. The staff will determine compliance with the requirements of 10 CFR Part 20 in these areas during the COL review.

## 12.3 Radiation Sources

### 12.3.1 Regulatory Criteria

The applicable regulatory criteria and guidance include the following:

- 10 CFR 20.1301, “Dose Limits for Individual Members of the Public,” and
  10 CFR 20.1302, “Compliance with Dose Limits for Individual Members of the Public”

- 10 CFR 50.34a, “Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents—Nuclear Power Reactors”

- 10 CFR 50.34(f)(2)(vii), as it relates to the conduct of radiation and shielding design reviews of spaces around systems that may contain accident source term radioactive materials.

- 10 CFR Part 20, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage.”

- 10 CFR Part 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low as Is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.”

- 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19, “Control Room”

- 10 CFR Part 50, Appendix A, GDC 60, “Control of Releases of Radioactive Materials to the Environment,” in Appendix A, “General Design Criteria for Nuclear Power Plants.”

- 10 CFR Part 50, Appendix A, GDC 61, “Fuel Storage and Handling and Radioactivity Control”

- RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” November 1978.

- NUREG-0016, “BWR-GALE Code,” Revision 1, January 1979

- NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” February 1995

### 12.3.2 Summary of Technical Information

The applicant will use the contained source terms described in the DCD as the basis for the radiation design calculations (shielding and equipment qualification) and personnel dose assessment. The applicant will use the airborne radioactive source terms in the DCD for the design of ventilation systems and for assessing personnel dose. The staff reviewed the source terms in the DCD to ensure that the applicant had either committed to follow the guidelines of the regulatory guides and staff positions in Section 12.2 of the SRP or provided acceptable alternatives. Where the DCD adheres to these regulatory guides and staff positions, the staff can conclude that the design meets the relevant requirements of 10 CFR Part 20, and 10 CFR Part 50, Appendix A, GDC 61.

### 12.3.3 Staff Evaluation

The staff reviewed the descriptions of the radiation sources given in DCD Tier 2, Chapter 11, "Radioactive Waste Management," and DCD Tier 2, Section 12.2, "Radiation Sources," to assess completeness compared to the guidelines in RG 1.70 and the criteria in Section 12.2 of the SRP.

#### 12.3.3.1 Contained Sources

In DCD, Tier 2, Section 12.2.1, "Contained Sources," the applicant describes the shielding design source terms, including location and all pertinent quantitative source parameters, during normal full-power operation, shutdown, and design-basis accident events. These source terms are consistent with a BWR operating offgas rate of 100,000 microcuries/second (uCi/s) after a 30-minute delay. The source terms associated with systems and components carrying radioactively contaminated fluids were calculated consistent with the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS)-18.1, "Source Term Specification," 1999. Filters and ion exchange beds in such systems were assumed to contain their maximum radioactivity before filter backwash or resin exchange.

The activation product Nitrogen-16 (N-16) is the predominant radionuclide during plant operations because of its short half-life and energetic gamma emissions. Since the ESBWR design does not have reactor coolant recirculation loops, N-16 is somewhat less of a consideration for primary containment shielding design. However, during power operation of the ESBWR, N-16 activity is a factor in the radiation sources for the steam and condensate systems components located outside of primary containment. The fraction of N-16 produced in the reactor core that is released into steam depends on reactor water chemistry. Injecting hydrogen into the reactor coolant to minimize the potential for stress corrosion of piping and components in contact with the reactor coolant results in significant increases in the concentration of N-16 in BWR steam. Reducing the amount of hydrogen injection necessary, by pretreating the reactor system with noble metals, mitigates the N-16 increase. The N-16 source term used in the ESBWR design considers both hydrogen injection and noble metal treatment of the reactor system. The applicant used this elevated N-16 source term, which is five times the concentration of steam leaving the reactor vessel specified in ANSI-18.1, to calculate the annual skyshine contribution from N-16 at two typical site boundary distances. In both cases, the resulting annual dose from the N-16 skyshine from operation of the ESBWR is well below the 10 CFR 20.1301 dose limits.

The applicant used the design-basis source term values for the various radionuclides in determining the shielding design necessary to obtain the desired plant area radiation levels for the ESBWR. In arriving at the design-basis corrosion product activity levels for the ESBWR, the applicant used a set of values that are reasonably conservative relative to current operating plant experience.

In accordance with the criteria in Section 12.2 of the SRP, Section 12.2.2, "Airborne and Liquid Effluent Sources for Environmental Consideration," of DCD Tier 2 describes the large contained sources of radiation which are used as the basis for designing the radiation protection program and completing shield design calculations. These sources include the reactor core; the reactor water cleanup/shutdown cooling system; spent fuel and the fuel and auxiliary pools cooling system; the main steam and feedwater lines; the liquid, gaseous, and solid radwaste systems; and other miscellaneous sources. For each of these contained sources, the applicant provided either the source strength by energy group or the associated maximum activity levels listed by isotope. The DCD provides system layouts within rooms or cubicles, as well as information about the type and size of components in these systems.

As part of RAI 12.3-8, the staff asked the applicant to explain how the "before and after" dose rates listed in DCD Tier 2, Table 12.2-5, for various components of the CRD system were factored into the ESBWR design. After reviewing the applicant's response to RAI 12.3-8, the staff issued the following Supplement No. 1 to RAI 12.3-8:

> In GE's April 20, 2007, response to RAI 12.3-8, GEH included Table 12.2-5: Radioactive Sources in the Control Rod Drive System. The estimated gamma dose rate for the Rotating Ball Spindle "before cleaning" of "0.0E+00 mSv/hr" appears to be in error. It would appear that the "before cleaning" dose rate value for this component would be larger than the "after cleaning" dose rate value listed at "3.0E-01 mSv/hr." Please correct this apparent discrepancy.

Subject to resolution of RAI 12.3-8, Supplement No. 1, this remains **Open Item 12.3-8**.

Section 12.2 of the SRP also states that this section of the DCD should include descriptions of any radiation sources containing byproduct, source, and special nuclear materials. However, this section of the applicant's DCD did not describe those radiation sources needed to construct and operate an ESBWR plant. This lack of information was the basis for RAI 12.3-9. After review of the applicant's response to RAI 12.3-9, the staff issued the following Supplement No. 1 to RAI 12.3-9:

> RAI 12.3-9 asked the applicant to provide a description of any sources (*i.e.*, calibration sources) needed to construct and operate an ESBWR plant, or provide justification why this should be left to the COL applicant. To the extent that radiation protection features for these sources are provided for in the design (shielding, separate source rooms, etc.), they need to be addressed in the DCD. To the extent that these design features are to be provided in a COL, please identify this issue as a COL action item.

Subject to resolution of RAI 12.3-9, Supplement No. 1, this remains **Open Item 12.3-9**.

In DCD Tier 2, Section 12.2.4, "COL Information," the applicant stated that the COL applicant will determine exact placement and duration of residence for the Cf-252 startup source and holder in the spent fuel pool (**COL Action Item 12.2.4.1**).

The ESBWR core activity release model for a core melt accident is based on the source term model from NUREG-1465. The use of the NUREG-1465 source term model complies with the requirements of 10 CFR 50.34(f)(2)(vii). Therefore, the staff finds the use of this accident source term acceptable.

In RAI 12.3-10 (listed below), the staff asked the applicant to verify that the source term assumptions in NUREG-1465 were used to determine the in-plant post-accident source terms and to provide the source term assumptions used in determining the dose rates indicated on the post-accident radiation zone maps in Section 12.3 of the DCD. In the applicant's response to this RAI, the applicant did not verify that the post-accident dose rates shown in DCD Tier 2, Figures 12.3-43 through 12.3-51 incorporate the source term assumptions in NUREG-1465.

> Verify that the source term assumptions in NUREG-1465, and the associated dose criteria in GDC-19, were used to determine the in-plant post accident source terms and resultant doses to plant personnel. Provide the source term assumptions used in determining the dose rates indicated on the post-accident radiation zone maps (DCD Tier 2, Figures 12.3-43 through 12.3-51).

Subject to resolution of RAI 12.3-10, this remains **Open Item 12.3-10**.

12.3.3.2 Airborne and Liquid Effluent Source Terms and Doses

The staff reviewed DCD, Tier 2, Revision 3, Section 12.2.2, in accordance with the guidance and acceptance criteria provided in SRP Sections 11.2 and 11.3. The staff's evaluation addressed compliance with the requirements of 10 CFR 20.1301 and 10 CFR 20.1302 and the design objectives of Appendix I to 10 CFR Part 50 under Sections II.A, II.B, and II.C. Compliance with Section II.D of Appendix I to 10 CFR Part 50 is left to the COL applicant in evaluating the cost-effectiveness of liquid and gaseous effluent treatment systems.

In reviewing DCD, Tier 2, Revision 3, the staff could not confirm that the gaseous and liquid effluent radiological source terms, methodology, and assumptions used in estimating doses to members of the public, and gaseous and liquid effluent concentrations in unrestricted areas, were consistent with the guidance provided in Section 11.3, "Gaseous Waste Management System," of the SRP, Section 11.2, "Liquid Waste Management System," of the SRP, and associated regulatory guidance.

The staff asked the applicant to provide additional information addressing the basis of the radiological source terms and associated doses to members of the public. The applicant responded to the NRC's RAI, and the staff's evaluations of these responses are discussed below. Sections 11.2 and 11.3 of the SER, respectively, present the staff's evaluation of whether the designs of the liquid waste management system (LWMS) and gaseous waste management system (GWMS) are acceptable and meet the requirements of 10 CFR 20.1301 and 10 CFR 20.1302 and the design objectives in Appendix I to 10 CFR Part 50.

12.3.3.2.1 Airborne Effluent Releases

In reviewing Revision 1 of DCD Tier 2, the staff could not confirm that the gaseous effluent radiological source term, methodology, and assumptions used in estimating doses to members of the public, and gaseous effluent concentrations in unrestricted areas, were consistent with the guidance in Section 11.3 of the SER and associated regulatory guidance. The staff's evaluation addressed compliance with the requirements of 10 CFR 20.1301 and 10 CFR 20.1302, and the design objectives of Appendix I to 10 CFR Part 50 under Sections II.B and II.C. Section 11.3 of this report presents the staff's review of the GWMS, as it relates to the design requirements of 10 CFR 50.34a and GDC 60 and 61.

In reviewing DCD Tier 2, Revision 3, the staff found that some information remained insufficient to determine the acceptability of the applicant's analysis and results. The staff requested additional information. The applicant responded to the RAI, and the staff's evaluations of these responses are discussed below.

In RAI 12.2-9 (with its two supplements), the staff noted that it could not duplicate the estimates of annual airborne activity releases presented in DCD Tier 2, Revision 1, Table 12.2-16, using the information presented in DCD Tables 12.2-15, 11.1-1, 11.3-1, 10.4-2, and 9.4-1, including information provided by the applicant in response to RAI 11.1-3. The staff asked the applicant to address these issues and provide information describing all input parameters used with the BWR-GALE code. In DCD Tier 2, Revision 3, Section 12.2.2.1 and Table 12.2-15, the applicant submitted an updated source term for all gaseous effluent releases.

In its response, the applicant provided new information used in deriving the estimates of total airborne radioactivity releases. DCD Tier 2, Revision 3, Table 12.2-16, lists these estimates. The new information presents models, equations, and values for specific parameters, either given in the new information or extracted from NUREG-0016. Generally, the staff independently confirmed the approach and most results, except in a few instances where specific results could not be duplicated or clarifications are being requested because of specific assumptions or values used in the calculations. The following presents items for which the staff is seeking further clarification to resolve these outstanding issues:

(1) The adjustment factors for gaseous effluent source terms presented in equation (1) are based on a rated power level of 4590 megawatts (MW), while the basis of all radioactive source terms presented in DCD Table 11.1-3 is defined as 4500 MW. Similarly, the derivation of all liquid effluent source terms is based on 4500 MW (DCD Table 12.2-19a). Provide the justification for using a power rating of 102 percent for the estimation of gaseous effluent source terms.

(2) The derivation of C-14 activity released is based on 34,200 kilograms (kg) as the mass of water subject to neutron irradiation and production of C-14 (p. 3 of 5). This value is smaller than the one (39,000 kg) applied in NUREG-0016 for a generic plant rated at 3400 MW. Provide the justification for using a value of 34,200 kg for a plant rated at 4500 MW and designed with a larger reactor vessel.

(3) For equation (2), provide the justification for the value of 0.4 as the water flash fraction. The text supporting the use of this equation is silent on the basis of this value. Provide the information justifying this value.

(4) For equation (3), provide the information with which to derive the Ai/At ratio for noble gases listed in DCD Table 12.2-16. Indicate whether the ratios are based on steam concentration (uCi/g) or steam release rates (uCi/s at 30 min). Note that DCD Table 12.2-16 presents source term estimates for Kr-90 and Xe-139, but DCD Tables 11.1-2a and 11.1-2b do not list these two nuclides. Accordingly, update DCD Tables 11.1-2a and 11.1.2b so as to include Kr-90 and Xe-139.

(5) In support of equation (3), the derivation of noble gas activity released is based on a steam mass flow rate of $9.65 \times 10^{+6}$ kilograms/hour (kg/h), while the basis of

all radioactive source terms presented in DCD Table 11.1-3 is defined at a steam flow rate of 8.76 x $10^{+6}$ kg/h. Provide the justification for using a different steam flow rate in this equation.

(6) For equations (4) and (5), provide the justification for the value of 0.9 as the condensation removal factor. The text supporting the use of this equation is silent on the basis of this value. Provide the information justifying this value.

(7) In deriving the release rate of Ar-41, provide a justification for using the NUREG-0016 value of 40 uCi/s as a design basis and then adjusting it downward by a factor of 5 as a normal operational release rate. In light of the qualifier noted in NUREG-0016, an average release rate of 20 uCi/s (see Table 2-37) seems more appropriate in characterizing normal operations than the value of 11 uCi/s used in this calculation. Using the information presented by the applicant, the staff's estimate of the Ar-41 source term is twice that derived by the applicant, using a holdup time of 1.1 days in charcoal decay tanks.

(8) In deriving the release rate of Xe-133 and Xe-135, provide a justification for not adjusting the release rates by the ratios of the power levels (4500 vs. 3400 MW) and capacity factors (0.92 vs 0.80) of 1.35 and 1.15, respectively. Using the information presented in MFN-06-212, the staff estimates are correspondingly higher for Xe-133 and Xe-135 source terms after making such adjustments. Provide information with which to resolve this discrepancy.

(9) In confirming radioactivity release rates from the drywell via equation (5), the staff could not duplicate the results for particulates nuclides, but confirmed those for all radioiodines and tritium. For particulates, the staff's estimates are consistently higher by a factor of 1000 for the 24 nuclides that were checked. Provide information with which to resolve this discrepancy.

(10) In confirming radioactivity release rates from the drywell via equation (3), the staff could not duplicate the results for 13 of the 15 noble gases, excluding Kr-90 and Xe-139. The staff's estimates are both higher and lower than those provided by the applicant by factors ranging from 0.1 to nearly 270. See related issues noted in item (4) on the need for further clarification on derivation of Ai/At ratio for all listed noble gases. Provide information with which to resolve this discrepancy.

(11) A review of the information presented in MFN 06-212, Supplement 2, indicates that the enclosure presents key and important information supporting the basis, models, and assumptions used in deriving airborne effluent source terms. Regarding the development of the airborne effluent source terms, DCD Section 12.2.2.1 briefly states that "The methodology of NUREG-0016 was used in determining the annual airborne release values in Table 12.2-16." The staff's observation is that the models, assumptions, and parameters presented in MFN 06-212 cannot be inferred from NUREG-0016 alone. Accordingly, the staff requests that the enclosure to MFN 06-212, once revised to address the above noted issues, be appended to DCD Section 12.2 and that the text in DCD Section 12.2.2.1 refer the reader to this appendix for specific details and

information on the derivation of the airborne source terms. This approach would make the presentation of the supporting information about airborne effluents consistent with the corresponding details provided in the development of the source terms for liquid effluents.

(12) The ESBWR DCD should describe the performance requirements of adsorbent media for the eight main charcoal beds and two guard charcoal beds and for the charcoal filters used in building ventilation exhaust systems. The performance of adsorbent media should be consistent with the method used in demonstrating compliance with the requirements of 10 CFR 20.1301 and 20.1302, and Appendix I to 10 CFR Part 50, as described in DCD Revision 3, Sections 12.2.2.1 and 12.2.2.2. Please update DCD Tier 2, Table 11.3-1, "Offgas System Design Parameters," to specify the delay time for krypton and argon in addition to xenon which is already included, and Table 12.2-15, "Airborne Sources Calculation," to specify the charcoal filtration efficiency for radioactive iodine.

Subject to resolution of these collective issues, tracked as RAI 12.2-9, Supplement No. 2, this remains **Open Item 12.2-9**.

In Revision 3 of DCD Tier 2, Section 12.2.4.2, the COL action item is incomplete in demonstrating compliance with NRC regulations for airborne effluents. In addition to demonstrating compliance with the dose objectives of Sections II.B and II.C of Appendix I to 10 CFR Part 50, the COL applicant needs to demonstrate compliance with Section II.D of Appendix I to 10 CFR Part 50; airborne effluent concentration limits of Appendix B (Table 2, Column 1) to 10 CFR Part 20; and dose limits of 10 CFR 20.1301 and 10 CFR 20.1302 to members of the public. The staff also asked the applicant to update this COL action in DCD Tier 2 for the purpose of fully reflecting all applicable NRC regulations. The applicant has identified this as **COL Action Item 12.2.4.2** for airborne effluents, pending confirmation in DCD Tier 2, Revision 4. Therefore, this item becomes **Confirmatory Item 12.2-21.**.

The requirements of regulatory criteria and guidance, as they relate to sufficient detail to demonstrate that the equipment of the GWMS will support the design objectives of Appendix I to 10 CFR Part 50 under Sections II.B and II.C and gaseous effluent concentration limits of Appendix B (Table 2, Column 1) to 10 CFR Part 20, will be satisfied once the applicant adequately responds to all open and confirmatory items.

In addition, a COL applicant referring to the ESBWR certified design is responsible for ensuring that offsite doses to members of the public, based on site-specific parameters, comply with the design objectives of Appendix I to 10 CFR Part 50 for gaseous effluents under Sections II.B. and II.C, effluent concentration limits of Appendix B (Table 2, Column 1) to 10 CFR Part 20, and Section II.D of Appendix I to 10 CFR Part 50 in conducting a cost-benefit analysis of installed gaseous effluent treatment systems.

Due to the open item that remains to be resolved for this section, the staff was unable to finalize its conclusion regarding the acceptability that the GWMS (as a permanently installed system) includes the equipment necessary to control releases of radioactive materials in gaseous effluents in accordance with 10 CFR 20.1301 and 10 CFR 20.1302, Appendix I to

10 CFR Part 50, requirements of GDC 60 and 61, and requirements of 10 CFR 50.34a. Section 11.3 of this report presents the staff's evaluation of the GWMS.

12.3.3.2.2 Liquid Effluent Releases

In reviewing DCD Tier 2, Revision 1, the staff could not confirm that the liquid effluent radiological source term, methodology, and assumptions used in estimating doses to members of the public, and liquid effluent concentrations in unrestricted areas, were consistent with the guidance in Section 11.2 of the SRP and associated regulatory guidance. The staff's evaluation addressed compliance with the requirements of 10 CFR 20.1301 and 10 CFR 20.1302 and the design objectives of Appendix I to 10 CFR Part 50 under Section II.A. Section 11.2 of this report presents the staff's review of the LWMS, as it relates to the design requirements of 10 CFR 50.34a and GDC 60 and 61.

In reviewing DCD Tier 2, Revision 3, the staff found that some information remained insufficient to determine the acceptability of the applicant's analysis and results. This lack of information was the basis for RAI 12.2-15. After reviewing the applicant's response to RAI 12.2-15, the staff issued the following Supplement No. 1 to RAI 12.2-15:

> In RAIs 11.2.2-8, 12.2-10 (and its followup), and 12.2-15, the staff asked the applicant to provide discussions and assumptions describing offsite dose receptor locations, the rationale for the exposure pathways listed in DCD Tier 2, Revision 1, Table 12.2-20b, and a listing of all model parameters used in calculating doses using the methodology of the BWR-GALE code (NUREG-0016) and LADTAP II code (NUREG/CR-4013).
>
> Based on the information presented in DCD Tier 2, Revision 1, Tables 12.2-19a, 12.2-20a, 11.2-3, 11.2-4, 11.1-3, and 9.3-2, the staff could not duplicate, using the BWR-GALE code, the average annual liquid effluent concentrations and releases listed in Table 12.2-19b. In DCD Tier 2, Revision 3, Section 12.2.2.3 and Tables 12.2-19a and 12.2-19b, the applicant revised the estimate of the annual average source term and effluent concentrations released in liquid effluents. Using the updated information, the staff's evaluation confirmed the estimates of radionuclide concentrations in liquid effluents against the concentration limits of Appendix B (Table 2, Column 2) to 10 CFR Part 20. The evaluation compared the results listed for each radionuclide and collectively by using the sum-of-the-ratios under the unity rule.
>
> With the updated information, the staff's evaluation confirmed the estimates of annual radioactivity releases for all but 13 of the 46 radionuclides listed in Table 12.2-19b. The applicant's results were found to be higher than the staff's analysis for Np-239, Sr-90, Te-132, and Cs-137, with factors ranging from about 1.1 to 4.0. The applicant's results were found to be lower than the staff's analysis for Br-83, Ru-103, I-131, I-132, I-133, I-134, I-135, Cs-136, and H-3 with factors ranging from about 0.2 to 0.9. Accordingly, please provide supplemental information with which to resolve these differences and update the DCD.

Subject to resolution of RAI 12.2-15, Supplement No. 1, this remains **Open Item 12.2-15**.

In Revision 3 of DCD Tier 2, Section 12.2.4.3, the COL action item is incomplete in demonstrating compliance with NRC regulations for liquid effluents. In addition to demonstrating compliance with the dose objectives of Section II.A of Appendix I to 10 CFR Part 50, the COL applicant needs to also demonstrate compliance with Section II.D of Appendix I to 10 CFR Part 50; liquid effluent concentration limits of Appendix B (Table 2, Column 2) to 10 CFR Part 20; and the dose limits of 10 CFR 20.1301 and 10 CFR 20.1302 to members of the public. The staff asked the applicant to update this COL action in the DCD for the purpose of fully reflecting all applicable NRC regulations. The applicant has identified **COL Action Item 12.2.4.3** for liquid effluents, pending confirmation in DCD Tier 2, Revision 4. Therefore, this item becomes **Confirmatory Item 12.2-22.**

The requirements of regulatory criteria and guidance, as they relate to sufficient details to demonstrate that the equipment of the LWMS will support the design objectives of Appendix I to 10 CFR Part 50 under Section II.A and liquid effluent concentration limits of Appendix B (Table 2, Column 2) to 10 CFR Part 20, will be satisfied if the applicant adequately responds to all open and confirmatory items.

In addition, a COL applicant referring to the ESBWR certified design is responsible for ensuring that offsite doses to members of the public, based on site-specific parameters, comply with the design objectives of Appendix I to 10 CFR Part 50 for liquid effluents under Section II.A, effluent concentration limits of Appendix B (Table 2, Column 2) to 10 CFR Part 20, and Section II.D of Appendix I to 10 CFR Part 50 in conducting a cost-benefit analysis of installed liquid effluent treatment systems.

Due to the open item that remains to be resolved for this section the staff was unable to finalize its conclusion regarding the acceptability that the LWMS (as a permanently installed system and in combination with mobile processing systems) includes the equipment necessary to control releases of radioactive materials in liquid effluents in accordance with the requirements of 10 CFR 20.1301 and 10 CFR 20.1302, Appendix I to 10 CFR Part 50, GDC 60 and 61, and 10 CFR 50.34a. Section 11.2 of this report presents the staff's evaluation of the LWMS.

12.3.3.3 Airborne Radioactive Material Sources

In DCD Tier 2, Section 12.2.3, "Airborne Sources Onsite," the applicant described the sources of airborne radioactivity for the ESBWR reactor design and described actions taken to minimize radioactive airborne concentrations in various parts of the plant.

The main source of airborne activity in the reactor building during operation is leakage of primary coolant. The containment drywell is not accessible during normal operation, and during maintenance, the drywell air is purged before access is permitted. In reactor building areas outside the drywell, the ventilation system routes air from areas of lower potential airborne contamination (*i.e.*, corridors) to areas of higher potential airborne contamination (*i.e.*, equipment rooms).

During refueling, some of the sources of airborne activity typically are evaporation from reactor internals and fuel pool evaporation. Evaporation from reactor internals will be minimized by keeping surfaces of reactor internals (*i.e.*, the steam dryer and separator) wetted or covered when removed from the reactor vessel. Fuel pool evaporation will be minimized by lowering the temperatures in the fuel pools and using the fuel pool ventilation system to sweep the fuel pool surface to prevent pool releases from mixing with the area atmosphere. The applicant estimates that the resulting airborne concentrations in the reactor building will be below the limits established in Appendix B, Table 1, Column 3, to 10 CFR Part 20.

The source of airborne activity in the fuel building is the spent fuel storage pool and equipment areas. Similar to procedures in the reactor building, fuel pool evaporation will be minimized by lowering the temperature in the fuel pool and using the fuel pool ventilation system to sweep the fuel pool surface to prevent pool releases from mixing with the area atmosphere. The applicant estimates that the resulting airborne concentrations in the fuel building will be below the limits established in Appendix B, Table 1, Column 3, to 10 CFR Part 20.

The main potential source of airborne activity in the turbine building is leakage from valves on large lines carrying high-pressure steam. The design provides for collection of this leakage and its transport back to the condenser. By circulating air from areas of lower potential airborne contamination to areas of higher potential airborne contamination, the applicant plans to minimize sources of airborne radioactivity from equipment leakage in occupied areas. The applicant estimates that the resulting airborne concentrations in the turbine building will be below the limits established in Appendix B, Table 1, Column 3, to 10 CFR Part 20.

The corridors and routine access operating areas within the radwaste building are not expected to have significant airborne radioactivity levels. The vents from tanks in the radwaste building are vented directly to the building ventilation system. Pumps and valves for radioactive systems are located in separate compartments that are not normally occupied. The radwaste building ventilation system routes air from areas of lower potential airborne contamination to areas of higher potential airborne contamination. The applicant estimates that the resulting airborne concentrations in the radwaste building will be below the limits established in Appendix B, Table 1, Column 3, to 10 CFR Part 20.

The applicant uses airborne radioactive source terms in the design of ventilation systems and for personnel dose assessment. RG 1.70 states that Section 12.2 of DCD Tier 2 should include a tabulation of the calculated concentrations of airborne radioactive material, by nuclide, for areas normally occupied by operating personnel. Section 12.2.3 of DCD Tier 2 describes the assumptions and parameters used to determine the maximum expected airborne radioactivity concentration levels during normal operations in the reactor building, fuel building, turbine building, and radwaste building. The staff finds that this approach constitutes an acceptable basis for satisfying the requirements of Appendix B to 10 CFR Part 20.

### 12.3.4 Conclusions

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusion regarding acceptability. The staff finds it acceptable for the applicant to defer discussion of the material addressed by COL Action Items 12.2.4.1 and pending COL Action Items 12.2.4.2 and 12.2.4.3. The staff will determine compliance with these COL action items during the COL review.

## 12.4 Radiation Protection Design

### 12.4.1 Regulatory Criteria

The applicable regulatory criteria and guidance include the following:

- 10 CFR Part 20
- 10 CFR 50.34(f)(2), as it relates to TMI-related requirements.
- 10 CFR 50.68, “Criticality Accident Requirements”
- 10 CFR 70.24, “Criticality Accident Requirements”
- 10 CFR Part 50, Appendix A, GDC 19, “Control Room”
- 10 CFR Part 50, Appendix A, GDC 64, “Monitoring Radioactivity Releases”
- RG 1.69, “Concrete Radiation Shields for Nuclear Power Plants,” December 1973
- RG 1.70
- RG 1.97, “Instrumentation for Light-Water-Cooled Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Revision 3, May 1983
- RG 8.2
- RG 8.8
- NUREG-0737, “Clarification of TMI Action Plan Requirements,” November 1980

### 12.4.2 Summary of Technical Information

The purpose of this review was to ensure that the applicant had either committed to follow the guidelines of the regulatory guides and applicable staff positions or offered acceptable alternatives for facility design features, shielding, ventilation, and area and airborne radiation monitoring to maintain occupational radiation exposures ALARA. Where the DCD adheres to these regulatory guides and staff positions, the staff can conclude that the design meets the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.” The following sections present the staff's findings.

### 12.4.3 Staff Evaluation

The staff reviewed the facility design features, shielding, ventilation, and area and airborne radiation monitoring instrumentation contained in DCD Tier 2, Section 12.3, “Radiation Protection Design Features,” for adherence to the guidelines in RG 1.70 and the criteria in Section 12.3-12.4 of the SRP.

12.4.3.1 Facility Design Features

The ESBWR reactor design incorporates several features to help maintain occupational radiation exposures ALARA in accordance with the guidance in RG 8.8. These design features are founded on the ALARA design considerations described in Section 12.1 of DCD Tier 2, and discussed in Section 12.2.3.2 of this safety evaluation report.

The ESBWR natural circulation design eliminates the need for reactor coolant pumps and reactor coolant piping typically found in BWR designs. Maintenance and inspection of these components (and supporting activities, *i.e.*, insulation removal and replacement) are significant sources of occupational radiation exposure in operating nuclear power plants. The simpler design of the ESBWR also facilitates personnel access and equipment maintainability in the upper and lower drywells. Work platforms are also provided for accessibility to main steam isolation valves and other equipment requiring routine maintenance. The lower reactor head area is designed with a minimum of equipment interference to facilitate CRD mechanism access for maintenance. In addition, a trolley system provides transport of the CRDs from the lower drywell to a dedicated maintenance area with lower radiation levels.

Equipment and piping layout are designed to reduce the exposure of personnel required to inspect or maintain equipment. Major sources of radiation are located in separate cubicles from their associated piping and pumps, as well as from each other, to reduce personnel radiation exposure from these components during maintenance. Pumps located in radiation areas are designed to minimize the time required for maintenance. Quick-change cartridge-type seals on pumps and pumps with back pullout features that permit removal of the pump impeller or mechanical seals without disassembly of attached piping are used to minimize exposure time during pump maintenance. The configuration of piping surrounding pumps is designed to provide sufficient space for efficient pump maintenance. Heat exchangers are constructed of stainless steel or Cu/Ni tubes to minimize the possibility of failure and reduce maintenance requirements. Fill and drain fittings are provided on radioactive systems and components that facilitate system/component flushing to reduce radiation dose rates during maintenance.

The applicant stated that specific details as to precise equipment definition are not available at this time. The applicant also stated that the COL holder will address material selection of systems and components exposed to reactor coolant to maintain radiation exposures ALARA. The applicant identified this as **COL Action Item 12.3.7.1**.

Lighting is designed to provide sufficient illumination in radiation areas to allow quick and efficient surveillance and maintenance operations. To reduce the need for immediate replacement of defective bulbs, multiple lighting fixtures are provided in shielded cubicles. Incandescent lamps, which require less time for servicing, are the only type of lamp used within the primary containment, the main steam tunnel, and the refueling level of the reactor building.

The ESBWR design has many features to minimize the spread of contamination within the plant. Contaminated piping systems are welded, to the extent practical, to minimize leaks through screwed or flanged fittings. For systems containing highly radioactive fluids, drains are hard piped directly to equipment drain sumps so that contaminated fluid does not flow across the floor to a floor drain. Smooth epoxy-type coatings are employed to facilitate decontamination in the event of spills or leaks. Pump casing drains are employed on radioactive systems whenever possible to remove fluids from the pump prior to disassembly. In containment, a circular stand in the reactor vessel head laydown area prevents contamination from inside the reactor vessel cover from spreading to the outside of the cover when the cover is in its storage space. In addition, the applicant can plasticize the floor inside the stand and the area of the cover storage point to control potential contamination releases.

In addition to designing equipment to comply with ALARA guidelines, the ESBWR plant layout is designed to reduce personnel exposures. The design provides adequate work and laydown space at each inspection and maintenance station. In addition, it provides for rigging and lifting equipment to facilitate the removal, transport, or replacement of equipment and the use of portable shielding during maintenance activities. Adequate support services (*e.g.*, power, compressed air, water, ventilation, and communications) will be available at work stations. Floor drains with appropriately sloped floors are provided in shielded cubicles where the potential for spills exists. Valves associated with highly radioactive components will be separated from other components and located in shielded valve galleries. Major components in radioactive systems will be located in shielded compartments where practicable. To minimize radiation streaming through wall penetrations, the ESBWR design calls for shield wall penetration rooms with offsets between the radioactive source and the normally accessible areas.

Radioactive piping will be routed through shielded pipe chases or shielded equipment cubicles, wherever possible, to minimize personnel exposures. Some short feed-through sections of piping may be embedded in concrete. By limiting the length of embedded piping to short sections, to the extent practicable, the applicant will facilitate the dismantlement of the systems and the decommissioning of the facility, as required by 10 CFR 20.1406. The equipment and layout design features described above conform with the guidelines of RG 8.8 for maintaining occupational radiation exposures ALARA. Therefore, the staff finds these features acceptable. The equipment and layout design features described above conform with the guidelines of RG 8.8 for maintaining occupational radiation exposures ALARA. Therefore, the staff finds these features acceptable.

The ESBWR design also incorporates several features to minimize the buildup, transport, and deposition of activated corrosion products in the RCS and auxiliary systems. The DCD states that the ESBWR design will reduce or eliminate the use of materials containing cobalt that are in contact with reactor coolant, except in cases in which the use of these materials is necessary for reliability purposes. Stainless steel is used in portions of the system such as the reactor internal components and heat exchanger tubes where high corrosion resistance is required. The nickel content of the stainless steels is in the range of 9 to 10.5 percent and is controlled in accordance with applicable material specifications of the American Society of Mechanical Engineers. Cobalt content is controlled to less than 0.05 percent in the XM-19 alloy used in the CRDs. To the extent practicable, Colmonoy is used for hard facings of components in the core area as an alternative to Stellite and other high cobalt alloys.

The use of butt welds instead of sleeve-welded joints will minimize the potential for creating crud traps in the weld areas of piping for those systems carrying radioactive liquids. Tanks containing radioactive liquid will have drain pipes connected at the lowest part of the tank and convex or sloped-bottom designs to minimize radioactivity deposition. Pipes are seamless and are adequately sloped for avoiding stagnation. Piping configurations are designed to minimize the number of “dead legs” and low points in piping runs to avoid accumulation of radioactive crud and fluids in the line. Straight-through valve configurations are used where practical, to minimize crud traps and radiation exposure associated with maintenance on these valves. Valve packing and gasket material are selected for long operating life to minimize required maintenance. Valves have back seats to minimize the leakage through the packing. Equipment and piping containing radioactive materials will have provisions for draining and flushing. These design features, which are intended to minimize the buildup, transport, and deposition of activated corrosion products in the RCS and auxiliary systems, are based on the guidelines in RG 8.8 and are, therefore, acceptable.

The applicant provided the staff with detailed drawings of the ESBWR plant layout which indicate the nine radiation zones used in the plant design. These radiation zones serve as a basis for classifying occupancy and access restrictions for various areas within the plant during normal operations and accident conditions. On this basis, the applicant establishes the maximum design dose rates for each zone and uses these as input for shielding of the respective zones. On the basis of its review of the detailed zoning drawings, the staff concludes that the applicant's method of plant zoning, for normal operations, is consistent with the guidance in RG 1.70 and the SRP. Therefore, the staff finds this method acceptable.

Areas in which an individual would receive a dose in excess of 5 Sv (500 rem) within a period of 1 hour at 1 meter from a radiation source or 1 meter from any surface that the radiation penetrates are posted with "Very High Radiation Area" signs. Controlled access to "Very High Radiation Areas" is provided by the COL applicant. The applicant identified this as **COL Action Item 12.3.7.3**.

The radiation zone maps in this section of the DCD initially did not contain incremental zone designations for area dose rates above 100 millirem/hour (mrem/h). As part of RAI 12.4-4, the staff asked the applicant to identify all areas of the plant with dose rates greater than 100 rads/h. After reviewing the applicant's response to RAI 12.4-4, the staff issued the following Supplement No. 2 to RAI 12.4-4:

> In GE's June 7, 2007, response to RAI 12.4-4 S01, GEH revised the estimated radiation zone designations for several rooms depicted in Figure 12.3-19. The revised radiation zone designations for three of these rooms are still inconsistent with the zone designations listed in GE's initial response to RAI 12.4-4.
>
> In GE's initial response to RAI 12.4-4, the following rooms are listed as having anticipated dose rates of >100 rads/hr (Zone I) during normal operations: 6106, 6107, and 6161. In GE's response to RAI 12.4-4 S01, these rooms are designated as having the following dose rates: Rm 6106 (< 10 R/hr, zone G), 6107 and 6161(<100 rads/hr, Zone H). Please clarify these apparent zone designation inconsistencies.

Subject to resolution of RAI 12.4-4, Supplement No. 2, this remains **Open Item 12.4-4**.

In accordance with Section 12.2 of RG 1.70, the staff asked the applicant to provide the composition and thickness of each radiation shield depicted in DCD Tier 2, Figures 12.3-1 through 12.3-22. In response, the applicant provided a table listing the wall, ceiling, and floor thicknesses for the rooms with the most significant plant radiation sources. As part of its response, the applicant stated that special shielding features, if required, will be defined by the COL holder. In response to this reply, the staff issued the following Supplement No. 2 to RAI 12.4-6:

> In the applicant's June 7, 2007, response to RAI 12.4-6 S01, GEH stated that, "if required, special shielding features using other materials such as lead blankets, lead curtains, etc., will be defined later by the COL holder." To the extent that these design features are to be provided in a COL, they should be identified as COL Action Items in the DCD.

Subject to resolution of RAI 12.4-6, Supplement No. 2, this remains **Open Item 12.4-6**.

In its review of the plant layout figures in DCD Tier 2, Section 12.3, the staff noted that the fuel building equipment entry facility was designated as having "wash down bays." To ascertain the purpose of this area, the staff issued the following RAI 12.4-11:

> DCD Tier 2, Figures 1.1-1 and 12.3-4 indicate "wash down bays" in the fuel building equipment entry facility. Identify what equipment is intended to be washed down in this facility. If contaminated or potentially contaminated equipment is to be washed down in this facility, discuss the design features employed to minimize the spread of contamination (including the provision for collecting and disposal of wash down fluids).

Subject to resolution of RAI 12.4-11, this remains **Open Item 12.4-11**.

12.4.3.2 <u>Shielding</u>

The objective of the plant's radiation shielding is to minimize plant personnel and population exposures to radiation during normal operation (including AOOs and maintenance) and during accident conditions while maintaining a program of controlled personnel access to and occupancy of radiation areas. The ESBWR design also includes shielding, where required, to mitigate the possibility of radiation damage to materials.

The DCD states that radioactive components and piping will be separated from nonradioactive components and piping to minimize personnel exposure during maintenance and inspection activities. When radioactive piping must be routed through corridors or other low-radiation zones, shielded pipe chases are provided. Where applicable, pumps and other support equipment for components that contain radioactive material are separated from the more highly radioactive components by locating them outside the component cubicle in separate shielded cubicles. Shielded compartments have labyrinth entrances to minimize radiation streaming directly through access openings. Penetrations are located to preclude a direct line of sight from the radioactive source to adjacent occupied areas. In selected situations, provisions are made for shielding major radiation sources during inservice inspection to reduce exposure to inspection personnel. These shielding techniques comply with the guidelines contained in RG 8.8 for protecting plant personnel and the public against exposure from various sources of ionizing radiation in the plant. Therefore, the staff finds these techniques acceptable.

The design of the ESBWR radiation shielding applies RG 1.69, ANSI/ANS 6.4, "Nuclear Analysis and Design of Concrete Shielding for Nuclear Power Plants," and ANSI/ANS 6.4.2, "Radiation Shielding Materials."

In its review of the plant layout figures in DCD Tier 2, Section 12.3, the staff noted that the radwaste piping gallery housed both radwaste piping and electrical equipment. To ascertain whether shielding was provided between the radwaste piping and the electrical equipment, the staff issued the following RAI 12.4-16:

> DCD Tier 2, Figure 12.3-12 indicates that the radwaste piping gallery between the Turbine Building and the Radwaste Building also contains electrical equipment. Describe this electrical equipment, including the anticipated frequency of maintenance associated with it. Is shielding provided between the piping carrying radioactive fluids and this electrical equipment? If not, provide a justification why the current design is ALARA.

Subject to resolution of RAI 12.4-16, this remains **Open Item 12.4-16**.

In response to RAI 12.4-17, the applicant provided missing radiation zone designations for several rooms in the radwaste building. The staff noted an inconsistency in the applicant's response and issued the following Supplement No. 1 to RAI 12.4-17:

> In addition to other information requested in RAI 12.4-17, RAI 12.4-17 noted that Figure 12.3-20 is missing radiation zone designations for several rooms in the minus (-)2350 mm elevation of the Radwaste Building. Although GEH provided the missing radiation zone designations in their response to this RAI, the staff noted that the radiation zone designations provided in the RAI response (zones E/F) appear to differ from radiation zones shown on Figure 12.3-20 (zones E/C) for Room 6283. Please clarify these apparent zone designation inconsistencies.

Subject to resolution of RAI 12.4-17, Supplement No. 1, this remains **Open Item 12.4-17**.

In RAI 12.4-19, the staff asked for additional information regarding the adequacy of the shielding surrounding various sections of the inclined fuel transfer tube (IFTT) system. Upon review of the applicant's response to this RAI, the staff issued the following Supplement No. 1 to RAI 12.4-19:

> In its February 21, 2007, memo, the staff issued Supplement 1 to RAI 12.2-19 concerning the core burn-up values used for the fuel with respect to GE's shielding analysis. This supplemental RAI also applies to RAI 12.4-19. Upon further review of GE's response to RAI 12.4-19, the staff finds that it needs the following additional information regarding the IFTT:
>
> a. In your response to RAI 12.4-19, verify that all of the dose rate measurements are correct in light of the fact that some of the dose rates are given in units of mrem/h and some in mSv/h.

b. In Figure 9.1-2 there appear to be two areas where the embedded IFTT comes very close to potentially accessible areas. One of these areas is in Room 2400 (rail car bay) near the rail supports of the main crane in the fuel building (roughly at level +13570). The other is in Room 2400 near the lower part of the fuel handling machine opposite the trapezoidal room at elevation +4650. Describe what features (both physical and administrative) are in place to restrict personnel access to these two areas during fuel transfer operations. Provide the thickness of the concrete at the narrowest point between the IFTT and each of these two areas and provide the corresponding maximum dose rate at these points from a spent fuel assembly in the adjacent portion of the IFTT.

c. On Elevation 13570 mm (Figure 12.3-6) there appears to be a hallway between quadrants of General Area 1600 in the reactor building which passes by the IFTT which is embedded in the concrete wall to the south of this hallway. The note on Figure 12.3-6 states that this hallway is listed as radiation zone I (<500 rem/hr) during spent fuel transfer.

- Provide the minimum concrete thickness between the IFTT and this hallway.

- Provide the maximum dose rate at this point from a spent fuel assembly in the adjacent portion of the IFTT.

- Describe what features (both physical and administrative) are in place to restrict personnel access to this hallway when fuel is being transferred in the IFTT.

- Indicate where this hallway is located on Figure 9.1-2.

d. In your response to RAI 12.4-19, you mention access stairs to the crane in the fuel building. Describe where these stairs are located (list appropriate figure(s) showing location of the access stairs) with respect to the IFTT.

e. Figure 9.1-2 indicates that there is an access plug (elevation +4650 mm) to access the portion of the IFTT which runs through the trapezoidal room. State what plant layout figure shows this access plug entrance to the trapezoidal room (it does not seem to be shown on Figure 12.3-4) and describe the access route to reach this access plug.

Subject to resolution of RAI 12.4-19, Supplement No. 1, this remains **Open Item 12.4-19**.

Potentially lethal radiation exposures could occur in the vicinity of any unshielded portions of the fuel transfer tube when a spent fuel assembly passes through this tube during refueling operations. Rooms 18P2 and 1702 provide access to the unshielded portions of the inclined fuel transfer system (IFTS) for periodic inspections. A system of physical controls, interlocks, and annunciators controls personnel access to these rooms. The interlock system between the door locks, the main operation panel, and the control room prevents activation of the IFTS while the rooms are accessible. Audible alarms and flashing red lights are provided inside and outside any IFTS maintenance area to warn personnel of IFTS operation and the potential radiation hazard. In addition, radiation monitors that enunciate alarms both inside and outside each room provide continuous indication of the actual radiological conditions.

Section 12.3.2 of the SRP states that the applicant must describe how the shielding parameters were determined, including pertinent codes, assumptions, and techniques used in the shielding calculations. Table 12.3-1 of the ESBWR DCD describes the shielding codes used to determine the adequacy of the station shielding design. Specifically, the applicant stated that it used the point kernel shielding codes QADF and GGG to calculate most gamma dose rates throughout the ESBWR plant. In addition, the two-dimensional discrete ordinate transport code, DORT, and the point-kernel code, QAD CGGP 1.0, are listed. These are appropriate analytical computer codes, commonly employed in the design of radiation shielding for commercial nuclear power plants. Therefore, the staff finds the use of these shielding codes to be acceptable to evaluate the adequacy of the ESBWR station shielding design.

Due to open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability of the design of the ESBWR ventilation systems.

12.4.3.3 Ventilation

Chapter 9 of the DCD addresses the ESBWR ventilation systems, which are designed to provide adequate heating, cooling, and air supply to areas of the plant. The determination of the airborne concentrations of radionuclides within the plant serviced by these ventilation systems has been left to the COL applicant. Appendix 12.A to the DCD provides a methodology for determining the airborne concentrations in each room and cubicle, and Tier 1 of the DCD provides a specific ventilation inspection, test, analysis, and acceptance criterion (ITAAC).

The ESBWR ventilation systems are designed to protect personnel and equipment from extreme environmental conditions, and to ensure that personnel exposure to airborne radioactivity levels is minimized. Further, the design ensures that the dose to control room personnel during accident conditions will not exceed the limits specified in GDC 19. The following design objectives apply to all ESBWR building ventilation systems:

- The systems are designed to make airborne radiation exposures to plant personnel and releases to the environment ALARA. In order to achieve this objective, the applicant will follow the applicable guidance provided in RG 8.8.

- The concentrations of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance will be below the concentrations that define an airborne surveillance and maintenance will be below the concentrations that define an airborne radioactivity area in 10 CFR Part 20 during normal power operation.

The source of airborne radioactivity for a room or area is primarily from equipment leakage within the specified area. The ESBWR design incorporates the following features to minimize this leakage and thereby reduce the sources of airborne radioactivity.

- For all areas potentially having airborne radioactivity, the ventilation systems are designed such that during normal and maintenance operations, airflow between areas is always from an area of low potential contamination to an area of higher potential contamination.

- Negative or positive pressure is used appropriately in plant areas to prevent exfiltration or infiltration of possible airborne radioactive contamination, respectively.

- ESBWR equipment design includes provisions for limiting leaks or controlling the fluid that does leak. This includes piping the released fluid to the sumps and using drip pans with drains piped to the floor drains. For systems containing highly radioactive fluids, drains are hard piped directly to equipment drain sumps so that contaminated fluid does not flow across the floor to a floor drain.

- Systems containing radioactive fluids are welded, to the most practical extent, to reduce leakage through flanged or screwed connections.

The ESBWR ventilation systems incorporate the following design features to minimize personnel exposures.

- Major HVAC equipment is located in dedicated low radiation areas to minimize exposures to personnel maintaining this equipment.

- HVAC ducting is routed outside pipe chases and does not penetrate pipe chase walls (which would compromise the shielding around the piping).

- HVAC ducting penetrations through walls of shielded cubicles are located to minimize the effects of radiation streaming in adjacent areas.

- HVAC filters are provided with adequate space for maintenance activities, such as servicing and filter changeout. The particulate and HEPA filters can be bagged when being removed from the unit to minimize the spread of contamination. In order to minimize personnel exposures from radioactivity in the charcoal filters, these filters are allowed to decay to minimum levels and then they are removed by a pneumatic transfer system.

These design criteria adhere to the guidelines of RG 8.8 for maintaining doses ALARA and are acceptable.

RAI 12.4-23 asked the applicant to list the ESBWR ventilation systems designed to operate during accident conditions and provide the resulting radiation dose rates from these systems in adjacent areas during accident conditions. Since the applicant’s response was incomplete, the staff issued the following Supplement No. 1 to RAI 12.4-23:

> RAI 12.4-23 asked GEH to list the ESBWR ventilation systems designed to operate during accident conditions and to indicate their location on plant layout drawings. GEH was also asked to describe the maximum radiation source term in the filter or adsorption media, and give associated radiation dose rates in adjacent areas. Finally, they were to describe design features to ensure that the

radiation exposures resulting from maintenance (filter change out) of these systems is ALARA.

The information contained in the modifications made to Revision 3 of the DCD (Section 12.3.3.3 and Table 12.3-10) to address RAI 12.4-23 do not adequately respond to the staff's concerns.

Please address the following issues:

a. On the plant layout drawings, indicate the location of the reactor building (RB) HVAC filter units.

b. Include a table in the DCD similar to Table 12.3-10 which shows the dose rates in the RB HVAC filter units and adjacent rooms under accident conditions.

c. In Section 12.3.3.3 of the DCD, GEH states that the shielding wall thickness between the RB HVAC filter cubicles is sized so that the dose contribution in any cubicle from the filter in the adjacent one does not exceed 250 mSv/hr. Describe what maintenance (*i.e.*, filter change-out), if any, would be required on the RB HVAC filter units under accident conditions.

   If these units would have to be accessed following an accident to aid in the mitigation of or recovery from an accident, show that an operator would be able to perform the necessary operations on these units without exceeding the dose criteria of 50 mSv (5 rem) whole body, or its equivalent to any part of the body for the duration of the accident (per 10 CFR Part 50 and GDC 19, Control Room).

d. Modify Figure 12.3-47 to show the post-accident radiation zones in the vicinity of the control building emergency filter units on level 9060 of the control building.

Subject to resolution of RAI 12.4-23, Supplement No. 1, this remains **Open Item 12.4-23**.

To determine the location of and dose rates associated with the liquid filtration units for the ESBWR, the staff issued the following RAI 12.4-24:

> Indicate the location of the filtration units for the reactor building, the radwaste building, and the fuel building, on plant layout drawings. Describe the maximum radiation source term in the filter or adsorption media, for each and give associated radiation dose rates in adjacent areas. Describe design features to ensure that the radiation exposures resulting from maintenance (filter change out) of these systems is ALARA.

Subject to resolution of RAI 12.4-24, this remains **Open Item 12.4-24**.

Due to open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability of the design of the ESBWR ventilation systems.

12.4.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Section 12.3.4 of DCD Tier 2 addresses radiation monitoring in five categories. These are:

(1) area radiation monitors needed for accident situations per RG 1.97 and area monitors for normal operations for ALARA and to meet the criteria in ANSI/ANS 6.8.1, "Location and design criteria for area radiation monitoring systems for light water nuclear reactors," 1981.

(2) high-range containment monitors to meet the criteria specified in NUREG-0737 II.F.1 (10 CFR 50.34(f)(2) (xvii)(D))

(3) in-plant airborne radioactivity monitors

(4) effluent monitors

(5) instrumentation to monitor accidental criticality (per 10 CFR 70.24 or 10 CFR 50.68)

To determine which area monitor meets the criteria of RG 1.97 and which complies with the guidance in ANSI/ANS 6.8.1, the staff issued the following RAI 12.4-25:

> DCD Tier 2, Sections 12.3.4.1 and 12.3.4.2 describe the ESBWR area radiation monitoring (ARM) system. Tables 12.3-2 through 12.3-6 list the monitors with their locations provided on Figures 12.3-23 through 12.3-42. However, the information is unclear. Clearly indicate which RG 1.97 category and accident monitoring type variable each ARM is provided to meet and show that the range of each monitor is consistent with RG 1.97. For those ARMs not provided for accident monitoring, clearly demonstrate that they meet the guidance in ANSI/ANS 6.8.1, or provide a justification for an alternative.

Subject to resolution of RAI 12.4-25, this remains **Open Item 12.4-25**.

The area radiation monitoring system (ARMS) continuously measures, indicates, and records the gamma radiation levels at strategic locations throughout the plant except within the primary containment (which is monitored by the containment monitoring system). Monitor readings, alarm setpoints, and operating status of ARMS are indicated on control room displays. The ARMS is designed to provide early detection and warning for personnel to avoid unnecessary or inadvertent exposure to radiation and to ensure that occupational radiation exposures are maintained ALARA in accordance with the guidelines stipulated in RGs 8.2 and 8.8. To inform personnel of local dose rates in the area, area radiation monitors include a local readout and audible alarm in addition to readouts and alarms in the main control room. In addition, visible alarms are located outside each monitored area so that operating personnel can see them before entering the monitored area.

As supplemented by the criteria specified in Item II.F.1(3) of NUREG-0737, 10 CFR 50.34(f)(2)(xvii)(D) requires that each applicant provide instrumentation to measure, record, and read out in the control room the containment radiation intensity (high level). Item II.F.1(3) more specifically states that the reactor containment be equipped with two physically separate radiation monitoring systems that are capable of measuring up to $10^5$ grays/hour (Gy/h) ($10^7$ roentgen/hour (R/h)) in the containment following an accident.

Section 12.3.3 of DCD Tier 2 states that four gamma sensitive ion chambers are provided within the primary containment to monitor gamma rays during normal, abnormal, and accident conditions. Two redundant sensors are located in the drywell and two in the wetwell. The monitors will be located such that they are widely separated to provide independent measurements, with a large fraction of the containment volume considered in both the wetwell and drywell. In addition, the selection of the location will consider reasonable access for personnel to allow for replacement, maintenance, and calibration of this monitoring equipment. The range of each monitor covers 7 decades from 0.01 Gy/h (1 R/h) to $10^5$ Gy/h ($10^7$ R/h). Since **Open Item 12.4-25**, discussed above, has yet to be resolved, the staff was unable to finalize its conclusions regarding the acceptability that the design and qualification of these monitors complies with the guidelines of RG 1.97 and NUREG-0737, Item II.F.1(3).

The staff issued RAI 12.4-28 to ascertain whether the high-range containment monitors described in Section 12.3.4 of DCD Tier 2 meet the criteria of NUREG-0737, Item II.F.1(3) (as required by 10 CFR 50.34(f)(2)(xvii)) and follow the guidelines of RG 1.97. Upon reviewing the applicant's response to this RAI, the staff issued the following Supplement No. 1 to RAI 12.4-28:

> The original RAI noted that the second bullet under DCD Tier 2, Section 12.3.4 indicates that two redundant high range monitors are provided in the drywell and two in the wetwell "as required by RG 1.97." GEH was asked to verify that these monitors meet the criteria of NUREG-0737 II.F.1 as required by 10 CFR 50.34(f)(2)(xvii)(D) and to indicate the location of these monitors on the plant layout drawings.
>
> The staff is in need of additional information for RAI 12.4-28 concerning which revision to RG 1.97 is being used; noting that Revision 3 to RG 1.97 contains monitor ranges. Please provide this information.

Subject to resolution of RAI 12.4-28, Supplement No. 1, this remains **Open Item 12.4-28**.

In RAI 12.4-29, the staff asked the applicant to describe its in-plant airborne radiation monitoring system, including the location and detection capability of the airborne monitors. Since the applicant's response to this RAI did not specify detector locations, the staff asked for the following information in Supplement No. 1 to RAI 12.4-29:

> The original RAI response indicates monitoring and sampling points in "selected" locations. Please identify the locations or identify the intended criteria for selecting the locations.

Subject to resolution of RAI 12.4-29, Supplement No. 1, this remains **Open Item 12.4-29**.

The airborne radiation monitoring equipment will be placed in selected areas and ventilation systems to give plant operating personnel continuous information about the airborne radioactivity levels throughout the plant. When appropriate, the airborne radioactivity monitors are located upstream of the filter trains to monitor representative radioactivity concentrations from the areas being sampled. Section 12.3 of the SRP states that airborne radioactivity monitors shall be able to detect the time integrated change of the most limiting particulate and iodine species equivalent to those concentrations specified in Appendix B to 10 CFR Part 20 (one derived air concentration (DAC)) in each monitored plant area within 10 hours (*i.e.*, monitors should be sensitive enough to measure 10 DAC-hours). DCD Tier 2 states that airborne radioactivity monitors in plant areas that may be occupied by plant personnel will be capable of detecting 10 DAC-hours. The airborne radiation monitoring system, as described in the DCD, meets the scope of the postaccident monitoring requirements in GDC 64 and the guidance of RG 1.97. Therefore, the staff finds the airborne radiation monitoring system acceptable.

The Process Radiation Monitoring System (PRMS) continuously samples and monitors airborne radioactivity in effluent releases and ventilation air exhausts for noble gases, air particulates and halogens. Airborne contamination is sampled and monitored at the stack common discharge, in the off-gas releases, and in the ventilation exhaust from the reactor, radwaste, and turbine buildings. Airborne radioactivity samples will be periodically collected and analyzed for radioactivity. In addition, the applicant will use portable air samplers for compliance with 10 CFR Part 20 limits to check for airborne radioactivity in work areas prior to personnel entry where potential airborne radiation levels may exist that exceed allowable limits. The PRMS is described in greater detail in Section 11.5 of DCD Tier 2.

Section 12.3 of the SRP states that the DCD must provide the criteria and methods for obtaining representative in-plant airborne radioactivity concentrations in all work areas. Furthermore, 10 CFR 50.34(f)(2)(xxvii) (as supplemented by the criteria specified in Item III.D.3.3 of NUREG-0737) requires that each applicant provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. Item III.D.3.3 more specifically states that each applicant provide equipment and associated training and procedures for accurately determining the airborne iodine concentrations in areas within the facility where personnel may be present during an accident. The applicant has stated, in DCD Tier 2, Section 12.3.7, “Combined Licensee Information,” that the COL applicant will address the criteria and methods for obtaining representative measurements of radiological conditions, including airborne radioactivity concentrations in work areas (Item III.D.3.3 of NUREG-0737). The COL applicant will also address the use of portable instruments and the associated training and procedures to accurately determine the airborne concentrations in areas within the facility where plant personnel may be present during an accident. The applicant identified this issue as **COL Action Item 12.3.7.2**.

Both the process radiation monitors and area radiation monitors are located in the fuel storage and associated handling areas in order to detect excessive radiation levels. Process radiation monitors monitor ventilation paths from the fuel storage area and, in addition to isolating the appropriate ventilation path upon receipt of an indication of high radiation, provide indication and alarms to the operator. Area radiation monitors are provided in fuel storage areas to detect high radiation levels and provide visual and audible indication to operating personnel. The staff finds that the use and location of these radiation monitors satisfy the radiation monitoring requirements of 10 CFR 50.68(b)(6), and therefore they are acceptable.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the compliance of the area radiation and airborne radioactivity monitors with the applicable requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70, as well as with the personnel radiation protection guidelines of RGs 1.97, 8.2, and 8.8. These monitors are designed to monitor both area and airborne radioactivity levels in the plant to ensure that doses to plant personnel are maintained ALARA.

12.4.3.5 Postaccident Access

Section 12.3.5 of DCD Tier 2 lists the areas of the plant that may require access to aid in the mitigation of, or recovery from, the consequences of an accident (referred to as vital areas in NUREG-0737 Item II.B.2). Figures 12.3-43 through 12.3-51 also indicate these vital areas, along with their postaccident radiation zone designations.

As stated in 10 CFR 50.3(f)(2)(vii), an applicant must fulfill the following requirements:

- Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials.

- Design, as necessary, to permit adequate access to important areas and to protect safety equipment from the radiation environment.

Item II.B.2 of NUREG-0737 provides additional guidance on how an applicant can meet these requirements. Item II.B.2 states that an operator should be able to access any vital area, perform the necessary functions to aid in the mitigation or recovery from an accident, and exit the area without exceeding $5 \times 10^{-2}$ Sv (5 rem) to the whole body or $5 \times 10^{-1}$ Sv (50 rem) to the extremities (per GDC 19). The dose rate in areas requiring continuous occupancy should be less than $15 \times 10^{-5}$ Sv/h (15 mrem/h) averaged over 30 days. DCD Tier 2, Section 12.3.5 states that the doses to access all vital areas following an accident are within regulatory guidelines.

After reviewing the postaccident radiation zone maps provided in the radiation zone layout drawings in DCD Tier 2, Section 12.3, the staff issued the following RAI 12.4-31:

> The post-accident radiation zones on DCD Tier 2, Figures 12.3-43 through 12.3-51 are incomplete. Layout drawings are only provided for the “Nuclear Island” and then only the dose rates in the vital areas and “access pathways” are provided. Although the legends on these drawings go up to Zone I (>100 Rem/hr), with the exception of one area on Figure 12.3-51, no area greater than Zone F (1 Rem/hr) is indicated on any of the figures.
>
> Provide a complete set of post-accident radiation zone drawings. Identify on these drawings the location of: (1) those systems and components that contain post-accident materials outside of the primary containment listed under Item III.D.3.3 of DCD Tier 2, Table 1A-1; (2) each specific area (not just the general room) requiring access to mitigate the consequences of an accident listed under Item II.B.2 of DCD Tier 2, Table 1A-1 (including technical support

> center and health physics facilities); and (3) the personnel access routes to, and egress routes from, these areas (not just a listing of the general rooms and stairs).
>
> Provide a detailed description of personnel actions to be taken in each area, the significant radiation sources associated with each, and an analysis of the radiation “mission” dose received (including dose from access and egress).

Subject to resolution of RAI 12.4-31, this remains **Open Item 12.4-31**.

The staff issued RAIs 12.4-32 and 12.4-33 to ascertain the applicant’s criteria for determining the radiation zones in ESBWR vital areas. After review of the applicant’s response to these RAIs, the staff issued the following Supplement No. 1 to RAIs 12.4-32 and 12.4-33:

> The last sentence of subsection 12.3.6 is not clear. The post-accident radiation zone maps should be based on the highest expected radiation dose rates under design basis accident conditions, as stated earlier in the subsection. The issues of whether the control room meets GDC 19, and that access to vital areas of the plant during accidents meet NUREG-0737 II.B.2 (50.34(f)(2)(vii)), or that the zone maps support the conclusions, is the subject of RAIs 12.4-31, 12.4-33 and 12.3-10. The response to 12.4-32 is incomplete as it refers the answer to RAI answers that have not been submitted.

Subject to resolution of RAIs 12.4-32 and 12.4-33, Supplement No. 1. These remain **Open Items 12.4-32 and 12.4-33**.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability that the information contained in Section 12.3.5 of DCD Tier 2 adequately addresses the relevant requirements of GDC 19 and 10 CFR 50.34(f)(2)(vii).

### 12.4.4 Conclusions

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability that the applicant has committed to follow the guidelines of the regulatory guides and staff positions in the applicable portions of Section 12.3-12.4 of the SRP. The staff finds it acceptable for the applicant to defer discussion of the material addressed by COL Action Items 12.3.7.1, 12.3.7.2, and 12.3.7.3.

## 12.5 Dose Assessment

### 12.5.1 Regulatory Criteria

The applicable regulatory criteria and guidance include the following:

- 10 CFR 20.1201, “Occupational Dose Limits for Adults”
- RG 1.70

- RG 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants—Design Stage Man-Rem Estimates," Revision 1, June 1979

### 12.5.2 Summary of Technical Information

The staff reviewed the applicant's dose assessment for the ESBWR facility contained in DCD Tier 2, Section 12.4, "Dose Assessment," for completeness against the guidelines in RG 1.70 and the criteria set forth in Section 12.3-12.4 of the SRP. The staff ensured that the applicant had either committed to follow the criteria of the applicable regulatory guides and staff positions in the applicable portions of Section 12.3-12.4 of the SRP, or provided acceptable alternatives. Where the DCD adheres to these regulatory guides and staff positions, the staff can conclude that the design meets the relevant requirements of 10 CFR Part 20. In addition, the staff selectively compared the applicant's dose assessment for specific functions and activities against the experience of operating BWRs. Radiation exposures to operating personnel shall not exceed the occupational dose limits specified in 10 CFR 20.1201.

### 12.5.3 Staff Evaluation

Section 12.3-12.4 of the SRP states that the applicant should describe any dose-reducing measures taken as a result of the dose assessment process for specific functions or activities. Section 12.4 of the DCD Tier 2 describes several dose-reducing measures and design modifications intended to reduce occupational exposures of plant personnel.

The reactor coolant system in the ESBWR is less complex than the reactor coolant systems in current BWR designs. The reactor coolant recirculation piping and pumps have been eliminated and a steel cylindrical shield has been provided around the reactor vessel to reduce drywell radiation fields. Since the recirculation lines are the most significant shutdown source of radiation in the drywell, removing the reactor coolant recirculation piping and pumps will have a significant effect on reducing the dose rates in the drywell outside the primary shield. This will reduce expected dose rates to personnel performing major drywell activities such as:

- Main Steam Isolation Valve (MSIV) Repair;
- Safety Relief Valve (SRV) Work;
- Fine Motion Control Rod Drive (FMCRD)/Automated Fixed In-Core Probe (AFIP) Work;
- Local Power Range Monitor (LPRM) Work;
- In-Service Inspection (ISI).

The ESBWR design includes provisions for planned access and work platforms for these major drywell activities to further reduce personnel doses. The ESBWR design replaces the conventional Traversing In-core Probe (TIP) system with fixed in-core detectors for calibrating the Local Power Range Monitors. This design eliminates maintenance, and resulting radiation exposure on the complex TIP drive and indexer mechanisms currently in use. In addition the potential radiation exposure associated with the TIP "backing out" events (i.e., the complete withdrawal from the reactor core of the freshly irradiated TIP probe into the drive housing) is eliminated.

Use of MSIV overhauling devices and an improved MSIV grinding system will result in an estimated 50 percent reduction in MSIV maintenance times. The ESBWR design provides overhead tracks and in-place removal equipment to facilitate handling and thereby lower personnel doses associated with SRV maintenance. Some of the features incorporated in the ESBWR design to reduce ISI maintenance and lower personnel doses include use of stand-off mirror type insulation around the reactor vessel, use of remote-operated mechanical devices for inspection of the RPV body and nozzle welds, and provision for additional ISI operations laydown space. The simplification of systems in the drywell has resulted in a significant reduction in the total number of valves and instrumentation in the drywell with a resulting decrease in maintenance time.

In the reactor building, equipment is more accessible which facilitates improved access control and improved equipment maintenance. Lifting points, monorails, and other installed devices are provided to facilitate equipment handling. Refueling exposures are decreased by use of an automated refueling platform, as well as a special stud tensioner for the RPV head bolts. Live-load valve packings are used to control valve stem leakage, thereby reducing valve maintenance and worker radiation exposures for valve repairs.

The condensate system in the ESBWR uses hollow-fiber filled filters which require approximately half the maintenance of typical systems. The condenser tube material is corrosion resistant which reduces leakage of corrosion products into the Condensate and Feedwater System.

As discussed above, the ESBWR design incorporates several improvements over current operating BWR designs. These improvements are intended to significantly reduce the personnel exposure associated with operational and maintenance activities. The occupational radiation exposure resulting from unscheduled repairs on valves, pumps, and other components will also be lower for the ESBWR than for current plant designs because of the reduced radiation fields, increased equipment reliability, reduced number of components relative to currently operating plants, improved water chemistry controls, and low cobalt usage.

Section 12.3-12.4 of the SRP states that the dose assessment will be acceptable if it documents in appropriate detail (including assumptions made and calculations used) the numbers and types of workers for each work activity, expected dose rates, and projected person-Sievert (person-rem) doses, in accordance with Regulatory Guide 8.19. In order for the staff to determine whether GEH conformed with the guidelines in RG 1.70 and the criteria set forth in Section 12.3-12.4 of the SRP, the staff issued a number of RAIs.

The applicant's dose estimates for plant workers were not consistent with the guidance in RG 8.19. On this basis, the staff issued the following RAI 12.5-1:

> Provide a complete tabulated dose assessment with a scope and detail consistent with the guidance in RG 8.19. Data should be presented in the format provided in RG 8.19 or an acceptable alternative. The analysis should clearly indicate the basis (*i.e.*, based on recent BWR experience or calculated based on similar tasks in other industries) for the staff-hour and dose rate estimates assumed, and show how each was adjusted to account for ESBWR specific design features. Estimates on work activities similar to the advanced boiling water reactor (ABWR) design (*i.e.*, control rod drive removal and maintenance) should be based on experience from operating ABWRs.

Subject to resolution of RAI 12.5-1, this remains **Open Item 12.5-1**.

After reviewing the applicant's dose assessment, the staff found that the average dose rate assumed for activities in the radwaste building appears low for these typically high-dose jobs. Therefore, the staff issued the following RAI 12.5-6:

> DCD Tier 2, Revision 1, Section 12.4.5 indicates that the radwaste building work activities considered in the dose assessment include movement of casks and liner, activated filter handling, resin moving and the removal of mobile radwaste processing skids. However, DCD Tier 2, Revision 1, Table 12.4-1 indicates that the average dose rate of 2.5 mrem/hr was assumed for these radwaste activities. Justify this low dose rate for what are typically high dose jobs.

Subject to resolution of RAI 12.5-6, this remains **Open Item 12.5-6**.

The applicant's dose assessment appears to assign a very low estimate of person-hours needed to maintain an operating ESBWR. This was the basis for the following RAI 12.5-8:

> DCD Tier 2, Table 12.4-1 gives an estimated total annual time of 33,131 person-hours to complete the radiologically significant work to operate and maintain an ESBWR. Exposure data reported to the NRC, and summarized in Volume 26 to NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and other Facilities," indicates that 35 U.S. BWRs reported dose records for a total of 59,991 workers (33,948 that received an annual dose of greater than 100 mrem) for 2004. An average BWR in 2004 had about 970 workers performing radiologically significant work.
>
> Assuming that similar numbers of workers will be required to operate and maintain an ESBWR, and that all the work included in Table 12.4-1 was completed by workers that would have an annual dose greater than 100 mrem, that translates into a work rate of only about 34 (33,131 person-hours per year divided by 970 workers) hours of radiological work per year per ESBWR radiation worker. Justify what appears to be a very low estimate of person-hours needed to maintain an operating ESBWR.

Subject to resolution of RAI 12.5-8, this remains **Open Item 12.5-8**.

On the basis of all of the design improvements and dose reduction features described above, GEH has estimated that the cumulative annual dose for operating an ESBWR plant will be 0.604 person-Sv (60.4 person-rem). This estimate is consistent with the Electric Power Research Institute design guideline of 1.0 person-SV (100 person-rem) per year and compares favorably with the average current BWR experience (the 2006 average collective dose for U.S. BWRs was 1.43 person-Sievert (143 person-rem)).

### 12.5.4 Conclusions

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability that the dose assessment for the ESBWR complies with the guidelines in RG 1.70, RG 8.19, and the criteria in the applicable portions of Section 12.3-12.4 of the SRP.

## 12.6 Operational Radiation Protection Program

### 12.6.1 Regulatory Criteria

The applicable regulatory criteria and guidance include the following:

- 10 CFR 20.1101
- 10 CFR 50.34(f)(2)(xxvii)
- RG 1.70
- NUREG-0737

### 12.6.2 Summary of Technical Information

Section 12.5 of RG 1.70 states that Section 12.5 of the DCD should contain a description of the applicant's operational radiation protection program. The applicant has stated that the COL applicant will be responsible for describing the operational radiation protection program. The staff will perform a detailed review of the applicant's operational radiation protection program against the criteria set forth in Section 12.5 of the SRP when it is provided at a later date by the COL applicant.

### 12.6.3 Staff Evaluation

The requirements in 10 CFR 20.1101 state that each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of 10 CFR Part 20. Section 12.5 of RG 1.70 and the SRP state that the operational aspects of an acceptable radiation protection program should address the following three areas:

(1) organization

(2) equipment, instrumentation, and facilities

(3) procedures

DCD Tier 2, Section 12.5, "Operational Radiation Protection Program," addresses the objectives and design of the ESBWR health physics facilities.

The stated objectives of the ESBWR design include health physics facilities and features that provide the capability for the administrative control of the following:

- the activities of plant personnel to maintain personnel exposure to radiation and radioactive materials ALARA and within the guidelines of 10 CFR Part 20

- effluent releases from the plant to maintain the releases ALARA and within the limits of 10 CFR Part 20 and the plant technical specifications

- waste shipments from the plant to meet applicable requirements for the shipment and receipt of the material at the storage or burial site

In order for the staff to be able to perform a more detailed review of these facilities and features, especially those located in the ESBWR service building, the staff issued the following RAI 12.6-1:

> DCD Tier 2, Section 12.5.2 discusses ESBWR facilities in the service building. Provide layout drawings (to the same scale as the other figures in DCD Tier 2, Section 12.3) of the service building, indicating the described facilities (including, but not limited to, the HP offices, control points, contamination control/monitoring stations, changing rooms (men's and women's), decontamination stations/showers, etc.). Indicate the designed plant access and egress control through these facilities.

Subject to resolution of RAI 12.6-1, this remains **Open Item 12.6-1**.

To obtain additional information regarding the purpose and radiation protection aspects of the shielded rooms in the health physics area, the staff issued the following RAI 12.6-2:

> DCD Tier 2, Section 12.5.2 states that shielded rooms are provided for radioactivity analysis and instrument calibration. Describe the radiation sources that these facilities are designed to contain, the shielding provided and any other protective considerations in the design. Does the ESBWR design provide a low background facility for personnel bioassay? If so, include a description with the above.

Subject to resolution of RAI 12.6-2, this remains **Open Item 12.6-2**.

Section 12.5 of DCD Tier 2 states that the health physics facilities are located in the services building. Access to radiologically controlled areas of the reactor, fuel, turbine, and radwaste buildings is normally through the entry/exit area of the health physics facilities. The health physics area contains the personnel contamination monitoring equipment, portable radiation survey instrumentation, decontamination shower facilities, and personnel changing rooms.

The applicant stated that it is the responsibility of the COL holder to describe fully the operational radiation protection program. The applicant identified the following three COL information items to describe the additional information to be provided by the COL applicant:

(1) **COL Action Item 12.5.4.1**—Radiation Protection Program

(2) **COL Action Item 12.5.4.2**—Equipment, Instrumentation, and Facilities

(3) **COL Action Item 12.5.4.3**—Compliance with Paragraph 50.34(f)(2)(xxvii) of 10 CFR Part 50 and NUREG-0737, Item III.D.3.3

### 12.6.4 Conclusions

As stated in Section 12.6.3 above, the COL applicant will be responsible for the description of the operational radiation protection program (per COL Action Items 12.5.4.1, 12.5.4.2, and 12.5.4.3) and will present the program for staff review at a later date as part of the COL application. The staff finds it acceptable for the applicant to defer discussion of the material addressed by these COL action items. When the COL applicant submits the operational radiation protection program, the staff will review it against the guidelines of the regulatory guides and staff positions in Section 12.5 of the SRP.

Due to the open items to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability of this section.

## 12.7 Minimization of Contamination

### 12.7.1 Regulatory Criteria

The applicable regulatory criteria and guidance include the following:

- 10 CFR 20.1406, “Minimization of Contamination.”
- NUREG/CR-3587, “Identification and Evaluation of Facilitation Techniques for Decommissioning Light Water Power Reactors,” June 1986.

### 12.7.2 Summary of Technical Information

Under the license termination provisions of Subpart E, “Radiological Criteria for License Termination,” of 10 CFR Part 20, 10 CFR 20.1406 requires, in part, that applicants for a new license describe how the facility design and procedures for operation will facilitate eventual decommissioning of the facility by minimizing, to the extent practicable, contamination of the facility and the environment, and the quantities of radioactive wastes generated. The staff reviewed the applicant’s contamination-minimizing design features contained in DCD Tier 2, Section 12.6, for completeness against the guidelines of 10 CFR 20.1406. The staff ensured that the applicant had either committed to follow the criteria of the applicable guidance or provided acceptable alternatives. Where the DCD adheres to these staff positions, the staff can conclude that the design meets the relevant requirements of 10 CFR 20.1406.

### 12.7.3 Staff Evaluation

Under the license termination provisions of Subpart E to 10 CFR Part 20, 10 CFR 20.1406 requires, in part, that applicants for a new license describe how the facility design and procedures for operation will facilitate eventual decommissioning of the facility by minimizing, to the extent practicable, contamination of the facility and the environment and the quantities of radioactive wastes generated.

NUREG/CR-3587, “Identification and Evaluation of Facilitation Techniques for Decommissioning Light Water Power Reactors,” issued June 1986, provides practical recommendations to facilitate the decommissioning of commercial light-water power reactors by reducing the radioactive exposures and waste volume generated during decommissioning activities. The report makes recommendations, based on actual decommissioning experience, applicable to the following three phases of plant life; decommissioning (end-of-life), plant operations, and plant design and construction.

Section 12.6.1 of the ESBWR DCD lists several design features intended to minimize contamination during plant operation to facilitate decommissioning, such as providing for the routing of radioactively contaminated piping through shielded pipe chases in lieu of embedding it in concrete, to the maximum extent practicable.

Section 12.6.2 of the ESBWR DCD addresses design procedures for operations that minimize the generation of radioactive waste.

To ascertain how the applicant incorporated the decontamination facilitation techniques of Section 5.2 of NUREG/CR-3587 into the ESBWR design, the staff issued the following RAI 12.7-1:

> Section 5.2 of NUREG/CR-3587 lists several decommissioning facilitation techniques that are applicable during the design and construction phase of a commercial nuclear power light water reactor. Describe to what extent each of these features were incorporated in the ESBWR design, or describe why the recommendation is not practical. Provide illustrative examples.

Subject to resolution of RAI 12.7-1, this remains **Open Item 12.7-1**.

The bulleted items in Section 12.6.2 of the ESBWR DCD did not adequately address how ESBWR design features minimize the generation of radioactive waste during decommissioning operations, as required by 10 CFR 20.1406. To ascertain this information, the staff requested the following in RAI 12.7-2:

> The discussions of the systems (liquid, solid, and gaseous, as well as waste management) provided in DCD Tier 2, Section 12.6.2 seem to be addressing minimization of effluents and solid waste from normal plant operation. Explain how the bulleted items (i.e., the segregation of wet and dry active waste for off-site shipment and burial) facilitate decommissioning operations. Describe how the ESBWR design minimizes the generation of radioactive waste during decommissioning operations.

Subject to resolution of RAI 12.7-2, this remains **Open Item 12.7-2**.

In RAI 12.7-3, the staff asked the applicant to identify any ESBWR piping or components that are below the grade of the plant site and that have a potential for leaking radioactively contaminated fluids. Since the applicant’s response was unacceptable, the staff issued the following Supplement No. 1 to RAI 12.7-3:

> The original RAI response indicates that the radwaste tunnel is designed to the same standard as the radwaste building, and that the radwaste building is designed to mitigate spills. What design features of these structures prevent

leakage from piping and components housed in them from reaching the ground water or environment for the life of the plant? Are these continuous pour, reinforced concrete structures, with no seams or joints? Are there expansion joints at the interfaces between the tunnels and the buildings? If so, how is leakage prevented through them for the life of the plant? Are expansion joints accessible for inspection and maintenance? Do the radwaste tunnels have design features to detect leakage (large acute, or small long term) from the systems into these tunnels? Is there any contaminated piping in the ESBWR design that will be buried in the ground, not routed through one of the radwaste tunnels? Does the Spent Fuel Pool (SFP) have a double liner with a tell-tale leak detection system? The additional information provided does need to be included in the DCD.

Subject to resolution of RAI 12.7-3, Supplement No. 1, this remains **Open Item 12.7-3**.

### 12.7.4 Conclusions

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusion regarding the acceptability that the applicant has committed to follow the guidelines in 10 CFR 20.1406.